Indian Point 3
Nuclear Power Plant
P.O. Box 215
Buchanan, New York 10511
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New York Power Authority

Robert J. Barrett
Site Executive Officer

October 15, 1997
IPN-97-143

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Licensee Event Report # 97-025-00
Automatic Reactor Trip due to a High Resistance Contact on a Reactor Protection Relay While Testing An Analog Channel

Dear Sir:

The attached Licensee Event Report (LER) 97-025-00 is hereby submitted as required by 10 CFR 50.73. This event is of the type defined in 10 CFR 50.73 (a)(2)(iv).

Also attached is the commitments made by the Authority in this LER.

Very truly yours,

Robert J. Barrett

Robert J. Barrett
Site Executive Officer
Indian Point 3 Nuclear Power Plant

IE22/1

Attachments

cc: See next page

cc: Mr. Hubert J. Miller
Regional Administrator
Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406-1415

INPO Record Center
700 Galleria Parkway
Atlanta, Georgia 30339-5957

U.S. Nuclear Regulatory Commission
Resident Inspectors' Office
Indian Point 3 Nuclear Power Plant

COMMITMENT LIST

Number	Commitment	Due
IPN-97-143-01	I&C Engineering will develop a preventative maintenance program for reactor protection relays. This will be completed by May 15, 1998, in time for implementation during the next refueling outage (RO-10). This outage is planned for 1999.	05/15/98
IPN-97-143-02	I&C will implement the reactor protection relay preventative maintenance program during the next refueling outage (RO-10). This outage is planned for the latter part of the year 1999.	09/1999

CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9710240038 DOC.DATE: 97/10/15 NOTARIZED: NO DOCKET #
FACIL:50-286 Indian Point Station, Unit 3, Power Authority of Stat 05000286
AUTH.NAME AUTHOR AFFILIATION
BARRETT,R.J. Power Authority of the State of New York (New York Power Au
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Forwards LER 97-025-00 re automatic reactor trip.Commitments made by util,encl.

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Indian Point 3	05000286	1 OF 4

TITLE (4) Automatic Reactor Trip due to a High Resistance Contact on a Reactor Protection Relay While Testing An Analog Channel

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
09	15	97	97	-025-	00	10	15	97	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): NA

POWER LEVEL (10): 030

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

20.402(b)	20.405(c)	✓ 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steve Manzione, I&C Engineering Supervisor	(914)736-8783

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	JC	RLY	W120	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE).	✓ NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On September 15, 1997 the plant was operating at approximately thirty percent reactor power. At 1623 hours, the plant experienced an inadvertent automatic reactor trip when the logic for two out of four channels for low pressurizer pressure was completed. One channel was placed in trip as per procedure in order perform instrument channel testing in accordance with Technical Specifications. At that time, the logic relay for another channel was already effectively tripped due to a high resistance contact causing an insufficient voltage being applied to the corresponding reactor trip relay. After the trip, the relay with the high resistance contact was replaced. Voltage drops across contacts on reactor protection relays were measured, relay contacts were cleaned and found to be acceptable. Review of the plant trip identified that other protective equipment responded as designed after the trip. Corrective action to preclude future similar events will be developed by engineering to include a preventative maintenance program for reactor protective relays. The event had no affect on the health and safety of the public.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Note: The Energy Industry Identification System Codes are identified within the brackets { }.

DESCRIPTION OF EVENT

On September 15, 1997 the plant was operating at approximately thirty percent reactor power, at 200 MWe and at normal reactor coolant {AB} pressure and temperature. At 1619 hours, Instrumentation and Control (I&C) Technicians commenced a surveillance test , 3PT-Q95, "Pressurizer Pressure Analog Functional Test." In order to test each pressurizer low pressure analog channel the test requires placing the channel in trip. This places a channel in a trip condition for each of the two reactor protection logic trains {JC}. At 1623 hours, when an analog channel was placed in trip, the plant experienced an inadvertent automatic reactor trip. Operators responded to the trip using emergency operating procedure E-0, "Reactor Trip or Safety Injection."

The plant protective equipment responded to the trip as expected. Rods {AA} inserted fully and auxiliary feed water pumps {BA} started. No safety injection actuation {JE} occurred nor was one required. At 1700 hours, when RCS average temperature reached approximately 540 degree F, operators closed main steam isolation valves {SB} and RCS temperature stabilized at a normal temperature of 547 degree F. Offsite power {EB} remained available during the event. The plant was maintained stable in the hot shutdown condition.

At 1747 hours, a non-emergency four-hour report (Incident Log No. 32930) for the reactor trip was made to the NRC Operations Center in accordance with 10CFR50.72(b)(2)(ii). At that time, the cause of the reactor trip was unknown. As a follow-up, on September 18, 1997, at 1503 hours, NYPA made a supplemental report for Incident Log No. 32930 to the Operations Center for this event to provide the cause of the trip.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

CAUSE OF EVENT

An investigation of the reactor protection circuitry disclosed that there was a reactor protection logic relay contact with high contact resistance. When the channel being tested was placed in trip for testing it opened select contacts in the train B trip matrix. These opened contacts in combination with the high resistance contact caused insufficient voltage to the corresponding reactor trip relay. Therefore, the two of four logic matrix was completed for train B causing reactor trip breaker B to open resulting in a reactor trip from train B. Only one contact had high resistance, therefore there was no indication to the operator or I&C technicians of the high resistance condition prior to the trip.

A contributing cause to the event was that cleaning reactor protection relay contacts was not a preventative maintenance activity.

CORRECTIVE ACTIONS

I&C replaced the relay that had the high resistance contact.

I&C functionally checked reactor protection relays in both logic trains by energizing and de-energizing their relay coil. Voltage across the reactor trip relay coils was monitored as relays were cycled. This provided indication of any voltage drop across the remaining closed contacts in the logic circuit. The results of this testing showed that the voltage drop across the contacts was up to one volt (2 cases), which was within the acceptance criteria. This was determined to be satisfactory when compared to the drop out voltage for the reactor trip relay. This testing effectively verified that approximately 400 relay contacts from approximately 200 relays were satisfactory. The relay contacts were cleaned using an electronic spray cleaner. The relays and associated contacts were found to function satisfactorily. The relay contacts were determined to be clean based on the voltage drop measurements.

FACILITY NAME (1)	DOCKET NUMBER	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL	REVISIO	
Indian Point 3	05000286	97	-025-	00	4 OF 4

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

CORRECTIVE ACTIONS, Continued

I&C performed an inspection of the safety injection logic relays (approximately 120). This inspection found these relays to be satisfactory.

I&C reviewed the failure history of these type of relays and found no indication that there is a set failure pattern associated with previous relay failures. Failures were found to be due to contact degradation or coil failure and were random in nature.

I&C Engineering will develop a preventative maintenance program for reactor protection relays. This will be completed by May 15, 1998, in time for implementation during the next refueling outage (RO-10). This outage is planned for 1999.

I&C will implement the reactor protection preventative maintenance program during the next refueling outage (RO-10). This outage is planned for the latter part of the year 1999.

ANALYSIS OF EVENT

The event is reportable under 10CFR50.73(a)(2)(iv) for an automatic reactor trip. A review of the past two years for similar reactor trips where protective electrical equipment contributed to the event identified LER 97-005-00, Manual Reactor Trip Initiated due to Overpower Delta T Channel Signal and Turbine Runback caused by a Foxboro bistable failure.

SAFETY SIGNIFICANCE

This event had no effect on the health and safety of the public. The reactor trip occurred as designed and no safety injection actuation occurred. If the plant was operating at full power, it is expected that the systems would respond as designed and safe shutdown of the plant would occur as it did in this event.